

NON-PUBLIC?: N  
ACCESSION #: 9010230119  
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Salem Generating Station - Unit 1 PAGE: 1 OF 07

DOCKET NUMBER: 05000272

TITLE: Reactor Trip on #13 S/G L-L Level Due to Personnel Error  
EVENT DATE: 09/10/90 LER #: 90-030-00 REPORT DATE: 10/09/90

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 079

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR  
SECTION:  
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:  
NAME: M. J. Pollack, LER Coordinator TELEPHONE: (609) 339-2022

COMPONENT FAILURE DESCRIPTION:  
CAUSE: B SYSTEM: TF COMPONENT: PSP MANUFACTURER: W120  
B TG SHV W120  
A JJ SRF W120  
REPORTABLE NPRDS: Y  
Y  
N

SUPPLEMENTAL REPORT EXPECTED: No

#### ABSTRACT:

On 9/10/90 at 1201 hours a reactor trip on No. 13 Steam Generator (S/G) Low-Low Level occurred. Prior to the event, the pipe between the No. 11TD900 valve and the main steam line sheared providing a steam flow path to atmosphere. To reduce the steam flow, the No. 11MS29 valve (MS Governor Valve) was closed. At 80% power, the 14MS29 valve is closed ("Partial arc control scheme"). Both the Nos. 11&14MS29 valves direct steam to the upper half of the Turbine. Therefore, with both valves closed, a significant dp across the HP Turbine developed. Contributing to this dp was opening the 11TD4 valve which resulted in bleeding steam away from the upper half of the HP Turbine. The Turbine shaft deflected creating an elliptical oscillation resulting in destruction of the Aux. speed sensor which generated an overspeed signal causing closure of the

MS29 valves. Closure of the MS29 valves led to the trip on No. 13 S/G Low-Low Level. The root cause of this event is attributed to personnel error. Ops. Dep't. management did not layout an approved plan of action in addressing the pipe break associated with the 11TD900 valve. Contributing factors were procedural inadequacy and inadequate training. This event has been reviewed by senior management. Those individuals involved in this event have been held accountable. Procedures have been revised to clearly identify the concerns with Turbine Valve Testing below 85% power. Operations Directive procedure OD-15 will be revised. An Operations Department "troubleshooting" procedure will be prepared. Damaged equipment was replaced. This event will be reviewed by the Nuclear Training Center for incorporation of lessons learned.

END OF ABSTRACT

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#### PLANT AND SYSTEM IDENTIFICATION:

Westinghouse - Pressurized Water Reactor

Energy Industry Identification System (EIIS) codes are identified in the text as {xx}

#### IDENTIFICATION OF OCCURRENCE:

Reactor Trip from 79% power on 13 Steam Generator Low-Low Level due to personnel error

Event Date: 9/10/90

Report Date: 10/09/90

This report was initiated by Incident Report No. 90-671.

#### CONDITIONS PRIOR TO OCCURRENCE:

Mode 1 Reactor Power 79% - Unit Load 900 MWe

#### DESCRIPTION OF OCCURRENCE:

On September 10, 1990 at 1201 hours, during normal power operation, a reactor trip on No. 13 Steam Generator (S/G) Low-Low Level occurred.

Prior to the event, the 1/2" Main Steam System {SB} pressure

indication line sheared upstream of the No. 11TD900 valve (Turbine Drain Instrument Isolation Valve). This provided a steam flow path to atmosphere. The 1/2" line is located on the No. 11 Steam Inlet Drain Line (1.5") between the 11MS29 valve (No. 11 Main Steamline Governor Valve) and the High Pressure (HP) Turbine. In an attempt to reduce the steam flow, to assess repair requirements without removing the Unit from service, the No. 11MS29 valve was closed. operations Procedure III-1.3.3, "Turbine Valve Tests" was used.

A "partial arc control scheme" is used in providing steam to the HP Turbine. In this scheme, the No. 14MS29 governor valve will remain closed until approximately 80% power is achieved. At 80% power, the valve is designed to begin opening to allow a power increase to 100%. When the No. 11MS29 valve was closed, the No. 14MS29 valve began to open; however, since both the No. 11 and No. 14MS29 valves direct steam to flow nozzles on the upper half of the HP Turbine, an unusual steam flow condition occurred. Also, Nos. 12MS29 and 13MS29 valves were opening more (to compensate for the closure of the 11MS29 valve). This condition resulted in a significant differential pressure (dp) across the upper and lower halves of the HP Turbine. Contributing to this dp was opening the 11TD4 valve (Air Operated Steam Inlet Drain Line Valve). It was opened as an attempt to reduce steam flow away from the 1/2" line break; however, it resulted in bleeding steam away from the upper half of the HP Turbine.

With the above conditions, the BP Turbine shaft deflected in the

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#### DESCRIPTION OF OCCURRENCE: (cont'd)

direction of the No. 11MS29 valve inlet nozzle. This deflection created a wave like action, traveling outward to the ends of the HP Turbine shaft. This wave was amplified towards the end of the shaft, due to shaft diameter reduction past the main bearing, resulting in an elliptical oscillation.

Located at the end of the shaft are three (3) speed sensors (Main, Auxiliary, and Spare) which are mounted on the Oil Pump approximately 40 mils off the shaft. The shaft elliptical oscillation resulted in damage to the Auxiliary speed sensor and the Main sensor. The Spare sensor remained undamaged. The damage to the Auxiliary speed sensor generated a 103% overspeed signal through the Electro-Hydraulic Control (EHC) electronics. The speed sensors work by producing pulses which are induced by the passing of the

shaft gear teeth. A single spike, exceeding the 103% overspeed protection control (OPC) setpoint (1854 rpm) occurred. The OPC signal then latched in for 5 seconds (by design) causing closure initiation (i.e., puffing) of all four (4) MS29 valves.

Control Room alarms received (in addition to the reactor trip alarms) included "EH Speed/Load Ch Failure" and "EH Prot Sys Trbl". Also, a speed monitor channel failure light was observed illuminated, indicating a difference of at least 55 rpm between the Main and Auxiliary speed sensors.

The closure of the MS29 valves resulted in a steam flow/feed flow mismatch which causes the BF19 valves (Feedwater Control Valves) to begin closing. The combination of the "shrink" phenomenon (caused by the closure of the MS29 valves) coupled with the reduction in feedwater flow lead to the reactor trip on No. 13 S/G Low-Low Level.

Following the reactor trip, a main steamline isolation {JE} was manually initiated (from Train B) to maintain Reactor Coolant System (RCS) {AB} Tavg (as per the Emergency Operating Procedure). Subsequently, the Unit was stabilized in Mode 3 (Hot Standby).

The Nuclear Regulatory Commission (NRC) was notified of the automatic actuation of the Reactor Protection System and the initiation of Main Steamline Isolation (an Engineered Safety Feature) in accordance with the Code of Federal Regulations 10CFR 50.72 (B) (2) (ii) on September 10, 1990 at 1233 hours.

#### APPARENT CAUSE OF OCCURRENCE:

The root cause of this event is attributed to personnel error. Operations Department management did not layout an approved plan of action in addressing the pipe break associated with the 11TD900 valve. Additionally, Operations Department management did not fully utilize available resources (e.g., System Engineering) in making the decision to proceed with Operations Procedure III-1.3.3, even though the reactor power level (79%) did not meet the procedural

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#### APPARENT CAUSE OF OCCURRENCE: (cont'd)

requirements for initial conditions (i.e., 85 to 90% power). Contributing factors to the cause of this event was procedural inadequacy and inadequate training.

Operations Procedure III-1.3.3 was found to be inadequate. The procedure Precautions section implied that the reason for requiring the plant to be between 85% and 90% power (Initial Conditions section), to perform the procedure, relates to unnecessary load reduction. Therefore, Operations management and shift supervision determined that it was acceptable to close the 11MS29 valve (per the procedure), in attempting to limit steam flow through the pipe break above the 11TD900 valve, even though the Unit was operating at 79% power. Engineering input was not fully utilized in making the decision to close the 11MS29 valve, using procedure III-1.3.3, or to open the 11TD4 valve.

A significant contributing factor to this event was Operator training inadequacy. Licensed Operator training includes training operators on Turbine operation. This training includes a discussion of the partial arc control scheme. Procedure III-1.3.3 is reviewed during this training. However, the reasons for the power restriction for use of the procedure is not addressed. It was found that the Westinghouse Turbine technical manual in the Nuclear Training Center library was not the latest revision (the 1981 version of the manual was found in the library). This earlier revision does not contain information relevant to load restriction. The latest revision contains a brief statement that identifies that Turbine valve testing below 85% power should not be conducted due to a potential for Turbine damage.

A factor which contributed to the inappropriate management decision to proceed with the use of Operations Procedure III-1.3.3 was lack of clear guidance as to the application of Operations Directive procedure OD-15, "Use of Operations Department Procedures". This procedure defines under what circumstances a procedure deviation from initial conditions can be applied. The guidance provided is not specific thereby allowing significant latitude in interpretation.

#### ANALYSIS OF OCCURRENCE:

The Low-Low S/G Level reactor trip prevents operation with the steam generator water level below the minimum volume required for adequate heat removal. The trip is actuated on two out of three low-low level signals in any S/G. The setpoint ensures adequate S/G inventory, at the time of a reactor trip, to allow for possible starting delays of the Auxiliary Feedwater Pumps {BA}; thus preventing S/G dryout and Reactor Coolant System {AB} thermal and hydraulic transients associated with a loss of the heat sink.

The OPC is designed to protect the HP Turbine from overspeed operation during conditions when the plant is not synchronized to the grid. The OPC is part of the EHC System {TG}. When an overspeed condition is sensed, the auxiliary governor emergency trip solenoids are designed

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#### ANALYSIS OF OCCURRENCE: (cont'd)

to release the operating fluid from the control emergency trip header. This will result in the closure of the control valves and the intercept valves for ten (10) seconds and five (5) seconds, respectively. The OPC also functions to electronically provide a close bias signal to decrease the turbine governor demand to zero volts. This closes all four (4) MS29 valves through their hydraulic control valve stopping steam flow in approximately 10 to 15 seconds.

Investigation found that the OPC auxiliary governor emergency trip solenoid valves would not function (i.e., close the turbine governor and intercept valves). However, the OPC did function by initiating a signal to decrease governor demand to zero volts thereby causing closure of the MS29 valves. The solenoid valves were original equipment.

The RPS functioned as designed and the heat sink was maintained during this event. Therefore, this event involved no undue risk to the health or safety of the public. However, due to the actuation of the RPS, this event is reportable in accordance with Code of Federal Regulations 10CFR 50.73 (a) (2) (iv).

The plant cooldown was greater than anticipated for a reactor trip from 79% power; however, it did not exceed the allowable Technical Specification limit. Tavg had decreased to approximately 532 degrees F when main steamline isolation was initiated. Emergency Operating Procedure EOP-TRIP-2 requires main steamline isolation if Tavg has not trended back to 547 degrees F approximately 10 to 15 minutes after a Unit trip.

During the plant cooldown, pressurizer level decreased to 14% and Pressurizer pressure to 1975 psig. Prior to the trip, Pressurizer level was 43% and Pressurizer pressure 2240 psig. The cooldown was compensated for by the manual initiation of Main Steamline Isolation (per the Emergency Operating Procedure).

The excessive cooldown has been attributed to operation of all three

(3) Auxiliary Feedwater (AFW) Pumps (2 motor driven and 1 steam driven). The motor driven and turbine driven AFW Pumps started as designed; i.e., loss of main feedwater, and all four Steam Generators had reached the low-low level setpoint. With all three (3) AFW Pumps operating, approximately twice the design basis AFW flowrate to the Steam Generators is realized (8.8E5 lbm/hr instead of 4.4E5 lbm/hr). The Turbine Driven Pump is designed to supply 100% flowrate and the Motor Driven Pumps are designed to provide 50% flowrate each.

#### CORRECTIVE ACTION:

This event has been reviewed by senior management. Those individuals involved in this event have been held accountable as applicable.

Procedure III-1.3.3 and other applicable procedures have been revised

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#### CORRECTIVE ACTION: (cont'd)

to clearly identify the concerns with Turbine Valve Testing below 85% power.

Operations Directive procedure OD-15 will be revised to clearly identify under what conditions the Nuclear Shift Supervisor can authorize deviation from a procedures Initiating Conditions.

An Operations Department procedure will be prepared which will define what actions shift personnel are required to take for abnormal plant conditions that require "troubleshooting" not addressed by existing procedures.

The main, auxiliary, and spare speed sensors were replaced.

The 11TD900 valve sheared piping was replaced. Preliminary analysis of the pipe has indicated that the pipe sheared due to vibration induced metal fatigue. This laboratory analysis is continuing. Upon completion of the analysis, the root cause of the pipe shear will be determined and appropriate corrective actions will be implemented.

An Engineering review of the required AFW System capacity was previously initiated due to similar cooldown rates experienced

following reactor trips (reference Unit 2 LER 311/90-029-00). This review is continuing.

This event will be reviewed by the Nuclear Training Center. Lessons learned will be incorporated into applicable training programs.

Investigation as to why the OPC solenoid valves did not function revealed that they were mechanically binding. Subsequently, the OPC solenoid valves and the emergency trip solenoid valve were replaced. The new valves were tested satisfactorily prior to the Unit synchronization on September 14, 1990. This testing included functional testing during the Unit restart.

A preventive maintenance program for the OPC solenoid valves and the emergency trip solenoid valve, which includes inspection and cleaning, has been established.

The Salem Unit 2 OPC solenoid valves and the emergency trip solenoid valve have been scheduled for replacement during the next outage of sufficient duration.

The Nuclear Training Center will audit the vendor technical manuals in the Nuclear Training Center Library to ensure that the latest revision is present. Additionally, a review of the program for ensuring that the latest revision of applicable technical manuals are obtained and controlled by the Nuclear Training Center Library will be conducted. Changes to the program will be made as applicable.

On September 13, 1990 at 0546 hours, the Unit entered Mode 1 operation. During startup, Turbine instrumentation was closely monitored. This instrumentation includes monitoring for

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CORRECTIVE ACTION: (cont'd)

eccentricity, bearing vibration, bearing metal temperature, return lube oil temperature, and main oil pump discharge pressure. The observed indications were compared with prior data. No abnormal indications were observed. Synchronization of the turbine-generator occurred on September 14, 1990 at 0519 hours.

General Manager -  
Salem Operations

MJP:pc

SORC Mtg. 90-139

ATTACHMENT 1 TO 9010230119 PAGE 1 OF 1

PSE&G

Public Service Electric and Gas Company

P. O. Box 236 Hancocks Bridge, New Jersey 08038

Salem Generating Station

October 09, 1990

U. S. Nuclear Regulatory Commission

Document Control Desk

Washington, DC 20555

Dear Sir:

SALEM GENERATING STATION

LICENSE NO. DPR-70

DOCKET NO. 50-272

UNIT NO. 1

LICENSEE EVENT REPORT 90-030-00

This Licensee Event Report is being submitted pursuant to the requirements of the Code of Federal Regulations 10CFR 50.73 (a) (2) (iv). This report is required within thirty (30) days of discovery.

Sincerely yours,

S. LaBruna

General Manager -

Salem Operations

MJP:pc

Distribution

The Energy People

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